Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/514
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

CENTRALE NUCLEAIRE DES ARDENNES
SPECIAL FEATURES AND IMPROVEMENTS COMPARED TO SIMILAR EXISTING PLANTS

P. ERKES, Bureau d'Etudes Nucléaires, Brussels.

P. GRAU, Electricité de France, Paris.

The "Société d'Energie Nucléaire franco-belge des Ardennes, S.E.N.A.", formed by "Electricité de France" on one side and the Belgian company "Centre & Sud" on the other, are presently building at Chooz, a village located 5 km from Givet, a large nuclear power plant. This plant is being built within the scope of an association contract with Euratom.

The power plant will be equipped with a PWR reactor of the "new generation type", feeding a single turbo-generator unit having a net electrical capacity of 266 MWe.

This reactor differs from its predecessors by a certain number of special features. Some of them are typical of the "new generation". They are essentially :

- chemical control,
- fuel bearing control rod followers,
- three regions core operation.

The other features are typical to the Centrale Nucléaire des Ardennes. They include mainly :

- the installation of the plant in a cavern inside the mountain,
- the construction of the equipment by French and Belgian manufacturers with the exception of the fuel core elements and of the absorbing section of the control rods,
- flow mixing grids within the fuel elements,
- in-core instrumentation,
- partial automation of plant control,
- safety injection by gravity,
- fuel transfer between reactor and storage pit with a new device.

These special features are discussed in the paper which concludes by a brief analysis of the plant cost and its economic perspectives.

Attached to the paper is a summary of the principal characteristics and a report on its present state of construction.

Translation provided by courtesy of the submitting Government

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/515
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

THE VULCAIN CORE POWER EXPERIMENT

by J. Storrer, Chef de Département
BelgoNucléaire, Brussels

S. Rigg, Deputy Chief Engineer
U.K.A.E.A., Risley

The Joint Anglo-Belgian Research and Development Programme of the Vulcain Reactor calls for the power experiment of a Vulcain core. This experiment will take place from 1965 to 1968 in the BR3 nuclear power plant at Mol, Belgium, on the site of the Centre d'Etude de l'Energie Nucléaire (C.E.N.).

The Vulcain reactor type is based on the variable moderation principle, using changing mixtures of light and heavy water, to compensate the decreases in reactivity due to fuel depletion. It permits a longer core life and power operation with all shut-off rods out of the core, leading to an improved neutron economy.

After discussing the objectives of the BR3/Vulcain Power Experiment the paper describes the core and the various modifications to be performed to the BR3 primary plant, followed by the proposed commissionning and operating programmes.

The main objectives of the experiment can be summarized as follows : (i) to verify the design parameters of a Vulcain core and its behaviour during the whole core life, (ii) to experiment the fuel behaviour at high burnups, (iii) to test various pieces of equipment.

The core consists of 91 theoretical cells of the same hexagonal cross section. 73 cells each contain one fuel assembly, while the remaining cells each accomodate one Zircaloy tube which may guide a circular flux-trap type absorber rod. The primary D_2O/H_2O mixture rises through the core almost totally in the fuel zone, a small part entering the Zircaloy "moderator" tubes. The fuel assemblies are developed and made half in the U.K. and half in Belgium, each manufacturer starting from UF_6 gas supplied by the U.K.A.E.A.

Translation provided by courtesy of the submitting Government

The modifications to the BR3 plant will take place during the latter part of 1964, after the first pressurized water core has been used up. They consist mainly in the adaptation of (i) the BR3 reactor vessel to the use of a Vulcain core and of (ii) the primary plant to the use of a D_2O/H_2O mixture.

The internal equipment of the reactor vessel is completely replaced. Ten absorbing rods are operated by the existing magnetic jack mechanisms located on the vessel lid, thus requiring special eccentric holding devices to connect eight of the rods which are offset. Four prototypes of hydraulic mechanisms will also be experimented. Extensive in-pile instrumentation is provided : (i) the flux wires slide inside thimbles attached to two vacated rod mechanism penetrations in the vessel lid (ii) the thermocouples and fission chambers are immersed and connected through a "collar" which is added between the vessel flanges.

The main problem concerning the primary systems is due to the use of a D_2O/H_2O mixture. It is indeed intended to operate BR3, which was originally designed as a H_2O plant, up to 2000 psia (140 kg/cm^2) and 500°F (260°C) which is higher than any other existing D_2O reactor. Extensive tests have therefore been performed on the BR3 plant to trace the leaks and determine their amount, in order to study the improvements to be effected to the circuits from that point of view.

The operating programme will comprise many cycling and perturbation tests to (i) subject the fuel to conditions similar to those encountered in a land based or ship propulsion unit, and (ii) obtain sufficient experimental data to arrive at a satisfactory mathematical model of the core behaviour. Various reactor control and water treatment methods will also be experimented, and the core will be operated at various pressures, down to the lowest pressure compatible with safe operation.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/516
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

"HERMES"

Fast superheater reactor

G.Tavernier, J.Chermanne, Cl.Descamps, M.Egleme, E.Fossoul, E.Jonckheere, J.Morelle, M.Stiévenart, L.Tollet & J.Van Dievoet.

BELGONUCLEAIRE, S.A. - Brussels, Belgium.

(Sessions 1.1. to 1.4., Power reactors, type f)

The growing interest in the use of gases for fast reactor cooling is well known.

One of the most promising gases is unquestionably the water steam which allows to consider a direct cycle. The use of dry steam in a reactor avoids the difficulties encountered in superheater reactor design. As known, these difficulties are mostly related to the presence of wet steam. Furthermore the steam can be fed into the turbine at a temperature corresponding to the one generally used in modern plants.

The dry steam can be produced, either by a classical reactor with direct cycle or indirect cycle through a steam generator, either through a Loeffler's cycle utilizing accordingly one single reactor as the thermal source for the plant.

Since a few years already, some studies have been performed at BelgoNucléaire in designing a reactor suitable for superheating steam produced in reactors of a well tested type,

Translation provided by courtesy of the submitting Government

These BelgoNucléaire studies are essentially based on a reactor of the "module" type. By "module", it is understood a pressure tube of relatively small diameter containing the fuel elements and used as a channel for the steam. These pressure tubes are juxtaposed. This pattern provides the necessary space between "modules" to install the detection and control devices for securing the plant operation.

The "modules" fit into a honeycomb structure in order to make their extraction easy for their possible replacement.

The reactor spectrum is essentially fast which allows use of appropriate structural materials, corrosion and radiation resistant at high temperature.

Since it is a fast reactor, a broad choice of cladding and dilution materials is also possible for the fuel elements. These fuel elements, depending on the design, could be clusters of elements made of uranium-plutonium mixed oxide, or spherical elements of the cermet type for instance.

The axial displacement of the fuel is possible. Partial loadings and unloadings are performed in order to optimize power distribution.

Among the advantages of the "module" pattern one can mention that the confinement of the fissile material is insured ipso facto in case of an accident and that the possible reactivity insertion during the partial or total water flooding of the core is easily controlled.

Beside the advantages related to the conception described, it is noteworthy to add those, well known, proper to the fast reactors and particularly their breeding ability.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/517
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

EXPERIMENTAL STUDY OF FLOW INVERSION IN THE BELGIAN ENGINEERING TEST REACTOR BR2

G.Stiennon
Head, Operating Group of the BR2 reactor and its auxiliary installations.

K.J.Bobin
Acting Reactor Manager BR2 (on loan from A.E.R.E.Research Reactors Division HARWELL).

H.Dopchie
Head, Department of Reactor Studies.

L.Fouarge
Engineer, Department of Reactor Studies.

F.Leonard
Head, BR2 Department.

F.Mathieu
Research and Information Division, EURATOM, Brussels.
Head, Instrumentation Section, Technology Department.

F.Motte
Head, Reactor Physics Section, Department of Reactor Studies.

D. Stoian
Engineer, Atomic Physics Institute, Bucharest (Attached BR2 under auspices of A.I.E.A.)

R.Vanhaelewyn
Head, Maintenance Section, BR2 Department.

Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium).

Work performed by the Association CEN-EURATOM for the exploitation of the BR2 reactor and its auxiliary installations (Contract 006/60/5 BRAB).

Translation provided by courtesy of the submitting Government

After a brief review of the reactor BR2, the Belgian high flux engineering test reactor, the paper describes the tests carried out recently to determine the maximum safe operating power. These tests consisted essentially in the measurement of the temperature of the surface of the fuel elements, using thermocouples imbedded in the cladding, at different reactor powers, and then deliberately inducing all possible fault conditions; these included the total loss of coolant flow (by stopping all main pumps, or by opening a by-pass valve or by closing the main flow valves) and loss of pressure in the primary circuit. Flux measurements were carried out in the core at the same time. The paper describes the organization of the tests and the equipment used.

During the tests, the maximum power attained was 37 MW, giving a maximum heat flux of 600 watts per square centimetre at the fuel element surface. This value is 1.5 times the planned operating figure, and is an exceptionally high value for this type of fuel element.

The corresponding thermal neutron flux (energy above 0.5 eV) in the fuel plates was 10^{15} $n.cm^{-2}.$ sec^{-1}. Even under these extreme conditions provided that pressure was maintained, the sudden loss of coolant followed by a flow inversion in the core did not lead to unacceptably high plate temperatures.

The paper gives an analysis of the results both at steady power and during the different tests, and in particular for the most severe transient.

After determination of the time of the inversion, the maximum "hot-spot" temperatures in the fuel are calculated. The upper limit of specific power is then determined so that boiling does not occur even in the hottest channel.

These data permit the safety assessment of this type of reactor at high specific powers and for different core configurations.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/518
BELGIUM
February 1964
Original: FRENCH

Confidential until official release during Conference

Abstract

PRACTICAL RESULTS AND PROSPECTS OF THE USE OF SOLUBLE POISON IN THE PRESSURIZED WATER REACTORS.

M. PLUMIER and P. STACQUEZ, Ateliers de Constructions Electriques de Charleroi.

P. DOZINEL, R. MEYLLE and M. PREAT, Bureau d'Etudes Nucléaires, S.A., Brussels.

G. BEUKEN, M. GUEBEN and L. MAESEN, Groupe Industriel pour l'Exploitation du BR3 du Centre d'Etude de l'Energie Nucléaire à Mol.

The control by chemical shim is going to play an important role in the economics of pressurized water reactor operation. The homogeneous poisoning of the primary circuit water, avoiding the perturbating action of the control rods authorizes a more uniform distribution of the neutron flux; it opens the way towards a large initial investment of reactivity, and that way leads to an increase of the core life time.

The chemical shim is provided in the pressurized water plants being presently erected of planned; it will compensate for the fuel burn-up and follow the slow transients, such as those resulting from the xenon effect. The combination of the neutronic and chemical properties of the boric acid designated it as poison.

The present paper covers, in the first part, the problems encountered during the power operation of the BR-3 plant, in presence of boron and, in the second part, the influence of this type of control on the design and the operation of the pressurized water type plants, as well as in the treatment of radioactive wastes.

A. Power operation of the BR-3 plant with boron in the primary loop.

After analog computer studies and preliminary tests proved that a power operation in presence of boron was safe, two series of tests were run

Translation provided by courtesy of the submitting Government

successively ; the first one with 125 ppm, the second one with 350 ppm of boron (in boric acid form) dissolved in the water. Most practical cases have been covered :

- power operation at fixed load,
- operation at reduced load,
- slow and fast load transients,
- slow reactivity transients ; especially the xenon effect was followedwithout rod displacement.

The report describes some of the encountered problems, and gives with comments, the results of the tests, showing the evolution of the primary loop parameters (reactivity, water chemistry, activities).

B. Influence on the circuits

The introduction of the chemical shim did not radically modify the auxiliary circuits of the reactors presently under design. However, it seems interesting to review the hydraulic circuits, as a whole,especially the purification and the waste treatment system.

Some solutions to the purification system problem are discussed in connection with the various methods of boric acid removal.

Furthermore, feed and bleed needed for boric acid dilution, increase the quantities of liquid wastes. The paper discusses the problem of the boric acid removal, either by concentration and separation (different methods are suitable : single effect evaporation or compression evaporation, demineralization, electrodialysis, precipitation ...), or by controlled rejection, after storage and dilution.

The economical aspect is also considered.

Third United Nations International Conference on the Peaceful Uses of Atomic Energy

A/CONF.28/A/519
AUSTRIA
February 1964
Original: ENGLISH

Confidential until official release during Conference

Abstract

Heat Transfer Characteristics of Helical Tube Bundles as used in Steam Generators of Gas-cooled Reactors

P.V. Gilli
Waagner-Biro A.G., Vienna

Helical tube bundles are being used for steam generators of gas-cooled reactors of the natural uranium, CO_2-cooled and the high temperature helium-cooled type. The steam generators (primary heat exchangers) of the U.K.A.E.A. High Temperature Reactor Project Dragon that have been designed and manufactured by the Waagner-Biro A.G. in co-operation with the Dragon Project are of particular interest since they are compact units of extremely high power density. Heat transferred per unit tube bundle volume amounts to 12.0 MWth/m^3, per internal casing volume to 9.02 MWth/m^3. Steam rate per unit volume is 23,000 kg/h,m^3 and 5,770 kg/h,m^3, respectively. These values compare favourably with other existing conventional or nuclear steam generators with gases as heat transfer media.

The high density of heat transfer and steam production of the Dragon boilers is due, amongst other reasons, to the somewhat novel compact tube arrangement. Each one of the six Dragon steam generator units consists of seven concentric cylinders of helical evaporator tubes in a multistart arrangement with numbers of starts increasing with helix diameter. Hence the angles of inclination as well as the heated lengths of the tubes are equal despite the large diameter differences of the seven helices. Thus flow resistance and mass flow per tube are equal.

Little published data is available on the heat transfer characteristics of bundles of plain helically-wound tubes in general and no data on the effects of the continuous change between in-line and staggered tube arrangement - which is a characteristic of the

design described - in particular. An experimental programme dealing with gas-side as well as steam/water-side heat transfer was therefore put in hand and is reported upon.

Gas-side heat transfer experiments were carried out under participation of Dragon Project. Results can be described in the range above $Re_f = 2000$ by the following formula:

$$St_f = 0.37\ Re_f^{-0.4}\ Pr_f^{-0.667}$$

where the subscript f refers to gas properties evaluated at the mean film temperature taken at the logarithmic mean of the heat transfer area and where the mass velocity term in the Reynolds Number refers to a nominal free gas flow area taken as the arithmetic mean between free area of in-line arrangement and free area of staggered tube arrangement based on transversal openings. A transition of flow regimes is apparent below $Re_f = 2000$ but no instabilities or oscillations were observed.

Due to the high heat flux in the steam generators of the Dragon reactor experiment (maximum values are of the order of 500 kW/m^2), steam/water side heat transfer is of special interest. Actual values of the boiling heat transfer coefficient were measured over the length and around the perimeter of the test tubes. For some cases, "burn-out" conditions were determined.